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OPERATIONAL EXPERIENCE OF THE FIRST ATOMIC POWER STATION AS AN EXPERIMENTAL SET – UP

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A great number of special experiments had to be when designing new nuclear power plants and in the first place the Kurchatov Atomic Power Station. These experiments included: a study of water boiling conditions and steam superheating ones in tubular fuel elements; a study of the formation, accumulation and release of radiolytic O_2 and H_2; a study of the water conditions; a study of the radioactive deposits from superheated steam; a study of the behaviour of graphite and steel in the pile.

However, the major problem in designing new atomic power plants was the experimental study of fuel elements. In order to solve all these problems it was necessary to build a number of experimental loops in the pile of the First Atomic Power Station.

§1. EXPERIMENTAL LOOPS

At present, tests are being conducted on the following experimental loops in the pile:
1. two-circuit steam-superheat loop;
2. natural-circulation water loop;
3. water unit for chemical experiments;
4. high-pressure water unit;
5. high-melting organic coolant loop;
6. low-melting organic coolant loop.

Following is a brief description of these loops.

TWO-CIRCUIT STEAM-SUPERHEAT LOOP

A study was made of the transient conditions on the loop, a study of the water conditions in radioactive coolant circuit, including problems of the formation, accumulation and release of radiolytic O_2 and H_2 and problems of deposits on fuel elements and turbine vanes.

A study was also made here of the behaviour of steel in contact with the coolant in its operating parameters. But the basic problems studied on that loop were testing fuel elements and certain structure elements of the fuel channels for the Kurchatov Atomic Power Station.

A schematic diagram of a two-circuit water loop with nuclear steam-superheating is shown in Fig. 1. The pipes and equipment are made of stainless steel. The loop has two radioactive circuits; its diagram is similar to the Kurchatov Atomic Power Station.

The coolant circulation in the primary circuit may be followed on the diagram. From the pressure pump the water is fed into the distributing collector after which it flows along separate evaporating channels. Water consumption is regulated with valves mounted on the inlets of the pipe channels. After the evaporating channels, the water or the steam-and-water mixture flows through separate circulation circuits and gathers in the general header then directed along a pipe into the evaporator. Then if need be, the water completely or partially goes through an additional cooler.

In the evaporator the heat of the primary circuit is wholly or partially transferred to the water of the secondary circuit. The secondary circuit is designed as an open circuit of circulation and makes it possible to conduct simultaneous tests of three steam superheating channels. With the aid of a plunger feed pump, the water of the secondary circuit is fed to the preheater (4) — regenerative heat exchanger where it is heated by cooling the superheated steam. Then, heated to the saturation temperature, the water is directed to the evaporator. Here it completely evaporates, due to the heat received from the primary circuit. The humid steam is further dried in the linear separator (2) and then in the header it is distributed along separate loops of the steam superheating channels. Each loop of the superheating channels has regulating valves and additional individual separators to remove moisture from the steam, which was formed in the supply loops, because of the decrease in pressure and heat lost through the walls of the pipes.

Superheated steam after the superheating channels is directed to the make-up water heater and may also be partially selected for experimental units (for example, on a stand of a running part of a turbine). The steam is finally condensed and cooled in the condenser and auxiliary cooler (5), and then dumped into the water tank (6) through the throttling device.

Any deviations detrimental to the operation of the unit (increase or decrease of pressure in the primary circuit, temperature increase in the fuel elements, increase of temperature in the superheated steam, increase or drop in water and steam consumption, deviations in the circulating pump) automatically stops the reactor by the safety trip system.

NATURAL – CIRCULATION WATER LOOP

Mention must be made of two peculiarities of the water loop with natural-circulation of coolants which make it possible to consider it as a highly prospective unit, to conduct ampoule tests of new fuel element compositions. Firstly, is its simple maintenance and reliability, and secondly, is its small dimensions which makes it possible to liquidate quickly the consequences of an accident caused by the destruction of the fuel elements.

The loop is assembled according to the double-circuit design (Fig. 2). The primary circuit is closed; it includes experimental ports (1), heat exchanger (2), volume compensator (3), inlet and outlet headers (4), sampling cooler (5).

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WATER — LOOP FOR CHEMICAL EXPERIMENTS

This loop is designed to study water chemistry, diffusion of nitrogen from volume compensators, corrosion of low-alloy steels, protection of these steels to cut down the rate of corrosion, testing filters to withdraw corrosion products from the circuit etc. A circuit diagram of the loop is shown in Fig. 3.

A regenerative heat exchanger (4) and an experimental port (2) are used to heat the coolants. Heat removal takes place in the heat exchanger (5) where the heat is conveyed to the water of the secondary circuit. There is a tank (6) with attachments to prepare the necessary solutions, ion-exchange filters (7) and sampling equipment to conduct chemical experiments in the circuit.

The thermo-mechanical equipment makes it possible to conduct experiments in temperatures of 50° to $300^\circ C$. The basic constructive peculiarity of this loop is that it has two parallel circuits: one of which is stainless, and the other — made of carbon steel. Such a system was used to get a more objective comparison of the speed of corrosion of various steels under similar conditions.

HIGH — PRESSURE WATER LOOP

This loop was built to check the various designs of fuel elements, to high burn-up fuel compositions, and to make a study of materials. Heat removal from the experimental ports was done by the water of the primary circuit, which was circulated by a glandless pump. The water of the primary circuit is cooled by the water of the secondary circuit in the heat exchanger, and the final heat removal, — by the circulating water from the general system. When conducting experiments, it is possible to change the water temperature at the channel inlets from 100 to $200^\circ C$. The water temperature is regulated at the channel inlets with a bypass line and valve on the primary circuit in the master heat exchanger. There is a probability of fission — fragment activity in the circuit if experiments are conducted in this loop. That is why all the basic equipment of the primary circuit is placed into a specially protected box. Counters are mounted on the channel exit pipes to control the dose capacity γ-radiation of the coolant leaving the reactor channel.

LOOPS FOR STUDYING ORGANIC LIQUIDS

It was necessary to build these loops in order to study the radiation resistance of organic coolants, their activation, the deposits of polymerization products on heating surfaces etc. It was also necessary to obtain some experience in operating a unit with organic coolants, to study the corrosion resistance of construction material in an organic medium under conditions of neutron radiation and to study the thermo-physical properties of the coolant during loop operation.

HIGH — MELTING ORGANIC COOLANT LOOP

The unit consists of the basic circulating circuit and a number of auxiliary systems. The coolant of the basic circuit, after passing through the hot channel where it undergoes heating

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and radiation, enters the heat exchanger and is cooled there with process water. From the heat exchanger, the coolant enters an expansion tank and from the tank it is pumped again to the channel. The unit has a charging tank where solid matter is melted down with subsequent re-pressuring into the expansion tank. Commercial nitrogen is used in the expansion tank and in the auxiliary units to create a blanket of inert gas.

An electric heating system is mounted on the pipes because the loop is designed to operate with organic matter having a melting point higher than the room temperature. The heat exchangers of the master circuit and the sampling coolers are heated with hot water.

LOW — MELTING ORGANIC COOLANT LOOP

The master circulating circuit loop includes: hot channel with fuel elements, circulating pumps, master heat exchanger, expansion deaerating tank, mechanical purification filter of the coolant, fittings and pipes.

There is a by-pass with a regulating valve in the main heat exchanger to regulate the temperature in the circuit.

The operating pressure is fixed in the circuit by feeding inert gas (nitrogen) into the gas volume of the expansion tank. The gas volume of the expansion tank is a receiver and collector of gases emitted from the coolant. The basic mass of the coolant is in the expansion tank. The required coolant level in the tank is maintained in the circuit with fresh coolant or dumping it into the drainage tank. An emergency cooling pump is automatically switched on if the main circulating pumps stop . This pump serves as a make-up and fills the circuit with fresh coolant which is preliminarily poured into the filling tank.

§2. GRAPHITE CORE OF REACTOR

The high temperature of the coolant and the large volume heat release in the fuel elements resulted in high temperatures of the graphite core. Nitrogen with an oxygen content of up to 0.2% in volume was added to the core at the APS in order to protect the graphite from burning up at high reactor temperatures.

A ten-year period of operation of the graphite core at temperatures up to 800°C and a high neutron flux showed that there was a low burn-up of the graphite in a nitrogen medium. It must be noted, that operating in nitrogen conditions filling the reactor core, there was a constant maintenance of oxygen, hydrogen and carbon dioxide content at a level of 1% in volume. A visual check of the surface of the graphite blocks in the reactor, and measuring the diameters of their holes at different points of the height of the core, made it possible to draw the conclusion, that the graphite core of the reactor at the APS after 10 years of operation was in a quite satisfactory condition.

Fig. 4 shows a chart of the diameter changes of the holes in one of the core cells along the reactor height, after ten years of core operation under conditions of neutron irradiation and high temperatures in a nitrogen medium.

The chart shows that there was a maximum decrease of the diameter holes of the graphite blocks in the centre of the core. The hole decreased from the nominal diameter of 65.0 mm to

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a diameter of 64.8 mm, i.e. by 0.1 mm along the radius., Such large deformations of the graphite blocks do not eliminate the designed gaps between the block and channel and that is why they are no hindrance to installing and removing channels from the reactor.

The integrated flux of fast and thermal neutrons along the centre of the core in a given cell was $0.5 \cdot 10^{22} n/cm^2$, and the graphite temperature on the inner surface of the block in operating conditions was 650^oC. The diameter of the holes was measured with an expecially designed meter with an automatic recording device. Taking into account the accuracy of the measurements, it can be considered that the volume changes in the graphite blocks after a long period of operation in the reactor of the First A.P.S. do not exceed 0.15%.

In order to determine the further increase of the operating temperatures of the graphite, a study was made of the graphite behaviour in a nitrogen medium at temperature intervals of 700°-1150^oC. The continuous method of weighing the sample in a gas flow was used to determine the speed of graphite oxidation.

In order to eliminate the influence of the gas flow velocity, the study was made in so-called "kinetic conditions", where the speed of oxidation reaction does not depend on the velocity of the gas flow (with consumption more than 200 1/hr).

For a quantitative determination of the process the specific speed of oxidation K_s, was used which is equal:

$$K_s = \frac{\Delta x}{S \Delta t} \qquad \frac{\Gamma}{cм^2 \ cек} \ ,$$

where
Δx — change of sample weight, grms.,
Δt — time of experiment, secs.,
S — surface of sample, cm^2.

The experiments were made on graphite samples irradiated with an integral dose $(0.4 + 0.8) \cdot 10^{21} P/cm^2$ (fast and thermal neutrons), in a medium of irradiated and unirradiated nitrogen in the reactor.

On the basis of the experiments a dependence of the specific velocity of oxidation to the temperature was drawn up which is shown in Fig.5. The chart shows, that by increasing the temperature from 700° to 1150^oC the oxidation speed in a nitrogen medium increases approximately ten-fold. Certain differences of the oxidation speed in irradiated and unirradiated nitrogen in temperatures of 700°-1000^oC may be explained by the distinctions in the integral doses of irradiation of the sample (as is known in practice, the higher the integral dose, the greater the oxidation speed).

It is interesting to note, that beginning with 1000^oC this difference in oxidation speed decreases and completely vanishes at 1150^oC. This phenomenon found in high temperatures, can be explained to a certain extent by the annealing of radiation damage in the graphite.

The initial nitrogen used in the experiments contained the following admixture:

oxygen 0.001	0.002%,
carbon dioxide	0.04 – 0.4%,

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| carbon monoxide | <0.04%, |
| hydrogen | 0.16—0.4. |

Graphite oxidizes in a nitrogen medium because of the presence in it of the above-mentioned admixtures.

Preliminary data does not make it possible to come to the conclusion of the possibility and advisability to use graphite cores in a nitrogen medium at temperatures over 800°C.

As steam can get into the nitrogen medium under certain operating conditions of graphite — water reactors, a study was made of graphite behaviour in a nitrogen medium with steam at high temperatures.

Studies were made to determine oxidation speed of irradiated graphite with a consumption of steam-gas medium of 9001/hr. The results of the experiments showed that in the interval of 700°-1000°C the oxidation speed increases by approximately 50% when increasing volume of steam concentrations by 5 to 20%, and then with a further increase of steam concentrations the speed remains nearly constant. It may be practically said, that with steam concentrations of over 20%, there are no essential increase in the speed of oxidation.

§3. CHANNELS AND TUBE FUEL ELEMENTS

The design of the channel and fuel elements of the pile, subjected to neutron radiation and high thermal loads underwent a very long test and proved itself very worthy. The design of the channel and the fuel elements have already been described by (1, 2). The design of the channel with tubular fuel elements is universal and so it may be used in boiling and in steam superheating operating conditions. No essential changes are required in the design of the channel for these operating conditions. All that is necessary is to add some resistance — orifice at the inlet sections of the fuel elements. One of the channels in the pile was operating in boiling conditions as far back as September 1956, then more than half of the channels were operating in boiling conditions by the middle of 1957 and then at the beginning of 1960 all channels were operating in these conditions. After boiling conditions were introduced, it was experimentally proved that tubular fuel elements could operate in steam superheating conditions. The most difficult task in the solution of this problem was the start-up conditions. To ensure proper cooling of the steam superheating channels when starting the reactor and gradually increasing pile capacity, method was tried; continuous transition from water cooling to steam cooling, with a gradual increase of steam content in the channels including at first, dry saturated steam and then superheated steam.

A channel of ordinary design successfully passed its tests of operating in steam superheated conditions for 1000 hrs., and over 50 outflows were made into steam superheating conditions, each time, changing over from water cooling to steam cooling. A more detailed description of channel operation in boiling and steam superheating conditions were given earlier by authors (1, 2, 3). Besides this, the channel may be used to operate in conditions of natural circulation on water under pressure, and also on boiling water. It was experimentally proved, that with gravity circulation, it was possible to receive 110kw thermal capacity in boiling and non-boiling conditions even with the least favourable gravity circulating loop.

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Gravity circulation is discussed by (4).

The basic technical specifications and operation of the channels and fuel elements of the First A.P.S. under various conditions are given in Table 1.

The design of the tubular fuel element, even if it breaks down, excludes the possibility of fission products and fissionable material entering the circulating loop because the pressure inside the loop is always higher than the gas pressure in the pile core.

Many years of operating experience confirmed this consideration and in so doing, outlined the prospects of applying tubular fuel elements in uniflow single-circuit loops.

The practice of fuel channel operation has brought about an essential increase in the designed lifespan of channel operation in the reactor.

Table 2 gives the data characterizing channel operation in the pile and the U burn-up fraction in them. As the table shows, the greater number of fuel elements (approx. 70% of total) operated in the pile 1.5-3.5 times more than the designed lifespan. There are also a number of fuel elements (approx. 3% of total) which operated in the reactor from 400 to 500 effective days.

Besides this, in order to determine the maximum allowable life of designed elements of the channel and fuel elements, the fuel element group (0.5% of total) underwent long tests for a duration of 600 effective days.

U^{235} burn-up was 30-32 Kg. per 1 ton and the calendar time of channel operation was approximately 44,000hrs. This burn-up was achieved by the wide application of the partial overload method and charging hot highly enriched channels (6.5-7%) into the pile.

During the operation of the reactor there were cases of loss of hermeticity of some hot channels but without a breakdown of the fuel elements. During the period of operation (1955-1964) the number of non-hermetic channels was only 0.3% of the total number of channels used. Practice has shown, that even with leakage of 100-1000gm/hr. the channels and the pile itself remain operationed and if need be ensure operation of the station at the required capacity.

A similar design of channels and fuel elements are used in the reactor of the Kurchatov Atomic Power Station.

Various tests of these fuel elements where made on experimental water loops of the pile at the First Atomic Power Station.

Table 3 gives some data on testing fuel elements of the Kurchatov Atomic Power Station.

§4. STUDY OF WATER AND SUPERHEATED STEAM RADIOLYSIS

There are three types of operating conditions characteristic of the Kurchatov Atomic Power Station which are: pile heat-up, transient conditions and superheated steam conditions.

1. Pile heat-up. When the pile is being warmed up, there is no boiling water. That is why one must not await the formation of a large amount of gases, because only when water boils, radiolytic hydrogen and oxygen volatilize from the aqueous phase and the recombination reaction is retarded.

2. Transient conditions. This condition of the pile operation is characterized by the beginning of water boiling with its gradual substitution into steam. During this period of operation

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the fuel elements are cooled by the steam-and-water mixture (X=99–1%). Also during this period there is a maximum evalution of gas as the recombination reaction is attenuated because of the gaseous product yield.

Some Data on Testing Fuel Elements of
Kurchatov Atomic Power Station

Table 3

Quantity of fuel elements in % of total tested	Production mW days/ton.	Heat flow $\frac{kcal}{m^2 hrs}$	Temperature of fuel element walls in °C
10	10000 – 15000	$1.3 – 1.6 \; 10^6$	400 – 430
30	6100 – 6500	$0.75 – 0.85 \; 10^6$	400 – 440
30	5500 – 6000	– ” –	– ” –
10	5000 – 5400	– ” –	– ” –
20	1000 – 5000	$0.5 – 0.8 \; 10^6$	430 – 550

3. Superheated steam condition. During this period there is a reduction of gas as the steam density is lower than the water or steam-and-water mixture densities.

During the experiments on the loop the main attention was put on the study of steam radiolysis under superheated condition, because this is the basic operating condition of the Kurchatov Atomic Power Station.

A balanced concentration of water radiolysis products is established during pile operation in the primary circuit of the loop (pressure 135 kg/cm^2, temperature at channel inlet 250°C, and at outlet 330°C). With power capacity of circuit at 1.1kw/kg, the H concentration did not exceed 2.5 n cm^3/kg and oxygen – 1.2 n cm^3/kg.

In the secondary circuit of the loop (pressure 40–80 kg/cm^2, steam temp at outlet 410°C) the yield of water radiolysis products greatly depends on the operating conditions of the loop and the concentration of disposable (dissolved) oxygen in the water.

Hydrogen leakage was observed in the system during the experiments which may be explained by the presence of sorption and diffusion through the walls of the hot channels as other reasons were excluded.

Hydrogen leakage increases as the wall temperatures are raised.

In the heat-up and transient conditions at water and steam temperatures up to 300°C, leakage can be 50% of the hydrogen released during radiolysis and during the superheated condition at a temp. of 400°C hydrogen leakage can be as high as 75–80%. The most probable leakage points are the channel, as its walls are 0.6–0.7 mm thick, and its welded seams.

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Yield radiolytic gases (hydrogen and oxygen) depending on operating
conditions and concentration of dissolved oxygen

Table 4

Operating conditions	Concentration of disposable (dissolved) oxygen mg/kg.	Radiolytic gas yield in ncm^3 per 100Kwh	
		Hydrogen	Oxygen
Heat-up	1.0 – 1.5	500–600	250–300
Transient	1.0 – 2.5	1660	830
Superheated	0.3 – 1.1	330–410	160–200
Superheated	0.01	2400	Oxygen – chemical bond.

§5. STUDY OF SUPERHEATED STEAM RADIOACTIVE IMPURITY DEPOSITS ON TURBINE VANE MODELS

The stand of the flowing part of the turbine has stationary vanes, similar to the flowing part of a BK-100 turbine in the section where there is the largest amount of deposits (from practice of convential energetics). Operational experience of industrial turbines shows that the chemical composition on stationary and mobile turbine vanes is approximately the same. The stand has three stages of models of flowing parts of BK-100 turbines operating with different steam parameters.

The first stage has a steam pressure of $30kg/cm^2$ and a temperature of 350°C.

The second stage has a steam pressure of $10kg/cm^2$ and a temperature of 250°C.

The third stage has a steam pressure of $2kg/cm^2$ and a temperature of 140°C.

The basic aim of the experiments was to determine the long-life radioactive isotopes which fall-out on models of the flowing parts of turbine vanes with the flow of superheated steam directly in the reactor channels. Experience has shown that there really was radioactive deposits

The γ- and β – radioactive deposits on vane models in stages after one month operation was as follows on the average:

first stage $2.7 \cdot 10^{-7}$ curie/cm^2,
second stage $2.0 \cdot 10^{-7}$ curie/cm^2,
third stage $4.0 \cdot 10^{-8}$ curie/cm^2,

with average superheated steam activity of 5.10^{-9} curie/kg.

Two days after shut-down 70% of the total radioactivity in the deposits desintegrated.

One month later, the residual radioactivity of the first two stages was 10% of the initial.

Long-life radioactive deposits on turbine vane models in conditions of operating the experimental stand were conditionally marked: Fe-59, Cr-51, Co-60.

Experiments showed that the greatest deposit activity was observed on the first two stages of the vane models with steam pressure of $30kg/cm^2$ and $12kg/cm^2$.

Deposition was mainly observed on the convex parts of the vane and were nearly absent on other parts of the vane.

A chemico-spectral analysis of the deposits revealed that the basic components of all deposits were: iron, nickel, calcium and cobalt was observed only radiometrically.

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Radioactive depositions are chemical stable compounds which are not affected by water and steam.

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BASIC TECHNICAL SPECIFICATIONS AND OPERATIONS OF CHANNELS AND FUEL
ELEMENTS OF THE FIRST ATOMIC POWER STATION UNDER VARIOUS CONDITIONS

Table 1

Basic technical specifications of channel and fuel elements

Dimensions of fuel elements in mm					Isotopic enrich-ment of $U235\%$	Total weight of U in chan-nel kg	Total surface of heat transfer channel m^2	Water ve-locity in channel m/secs.	Material
Height	Outside diameter	Thickness of outer shell	Inside diameter	Thickness of inner shell					
1700	14	0.2	8.2	0.4	5–7	4	0.175	3.5–4.5	Stainless steel

		Operating conditions				
Name of parameter	Dimensionality	Water	Boiling	Superheat	Gravity circulation	
					Non-boiling	Boiling
1	2	3	4	5	6	7
Max. temp. of fuel element walls	°C	340–350	340–350	470–500	360	340
Max. coolant consumption	kg/hrs.	1000–2000	700–1000	600	550	460
Effective pressure	kg/cm^2	100	100	90	100	100
Coolant at channel inlet	°C	180–190	190	t_s	–	–
Temperature at channel outlet	°C	270–310	t_s	370	<309.5	t_s
Heat load	$\dfrac{kcal}{m^2 hrs}$	0.8–1.8	10^6 $0,6-1,3 \cdot 10^6$	–	–	–
Thermal capacity of channel	KWT	–	–	–	110	117
Max. weight steam content at channel outlet	%	–	2 – 25	–	–	27
Dry residue in water	mg/kg	2–3	1.5–3	0.6–1.1	2.0–3	1.5–3
Water content — Ion concentration — PH		5.6–6.8	5.6–6.2	5.5	5.4–5.8	5.4–5.8
O_2	mg/kg	4.5–7.0	4.5–7.0	4.6	3–7	3–7
CO_2	"	0.5–1.0	0.5–1.0	0.9	0.5–1.0	0.5–1.0
NO_3'	"	0.1–0.7	0.1–0.7	0.1–0.15	0.1–0.7	0.1–0.7
Cl'	"	<0.02	<0.02	<0.02	<0.02	<0.02
CrO_4''	"	0.05–0.20	0.05–0.20	0.015	0.05–0.09	0.05–0.09

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		1		2	3	4	5	6	7
Water content	Ion concentration	Cr^{+++}	mg/kg	0.002–0.05	0.005–0.05	0.015	0.005–0.05	0.005–
		Fe	,,	0.05–0.1	0.05–0.1	0.05	0.05–0.1	0.05–
		Ni	,,	0.005–0.02	0.005–0.02	0.005	0.005–0.04	0.005–
		H_2	,,	0.4–0.7	0.4–0.8	0.3–0.5	–	–
		N_2	,,	5–12	5–12	5–12	–	~

Operation Time of Fuel Channels in Pile and U burn-up Fraction

Table

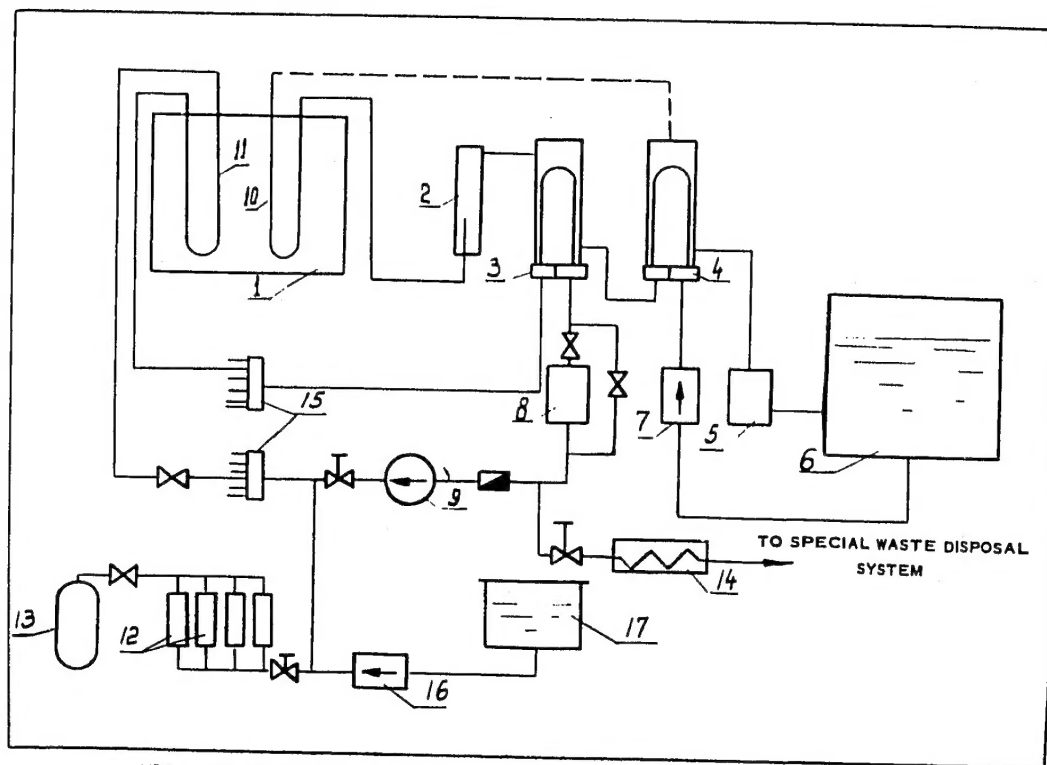| Number of effective days in operation	Average production mW days/ton	Average burn-up kg/t.	Quantity of fuel elements in % of total	Average burn-up fraction of U^{235} in % of its initial content	Note
600	28000–29000	30–32	0.5	60	U burn-up is considered as U^{235} consumption, a result of fission not taking into accoun radiative capture of U^{235}.
450	24400	27	0.2	53	
400	21200–23800	23.5–26.5	2.6	34–36	
350	18500–20900	20.5–23.2	6.5	30–32	
300	15200–17500	16.8–19.4	25.0	27–29	
250	12600–14900	14 –16.6	7.5	25–26.7	
200	10000–12500	11.1–13.3	8.5	21–26	
150	8600–9800	9.5–10.9	22.0	15–20	
100	5500–8300	6.1–9.2	25.2	11–14	
100	3000–5000	3.3–5.6	2.0	5–10	

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FIG.1. BASIC DIAGRAM OF A TWO-CIRCUIT STEAM-SUPERHEATED LOOP

1 — reactor; 2 — separator; 3 — evaporator; 4 — preheater; 5 — condenser; 6 — fed pump tank; 7 — feed pump; 8 — further coolant; 9 — circulating pump; 10 — steam channel; 11 — water channel; 12 — volume compensator; 13 — air condenser; 14 — shut-down safety cooler; 15 — headers; 16 — make-up pump; 17 — water tank;
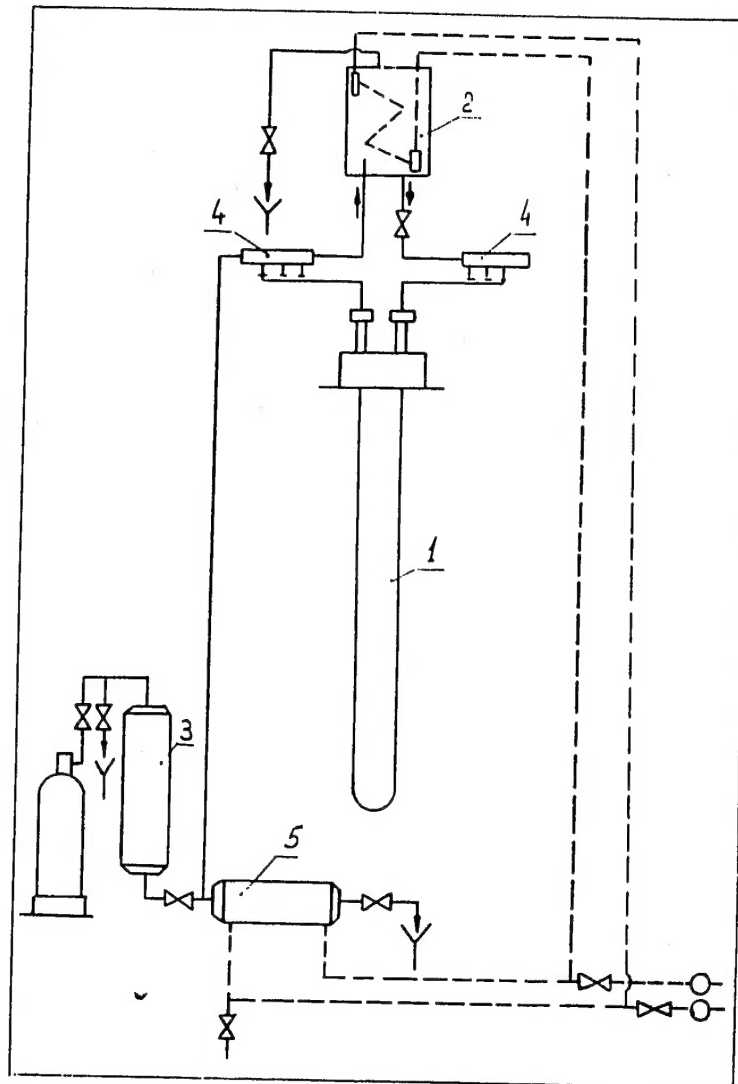
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TO SPECIAL WASTE DISPOSAL SYSTEM

FIG.2. WATER LOOP WITH NATURAL CIRCULATION
1 – experimental ports; 2 – heat exchanger; 3 – volume compensator;
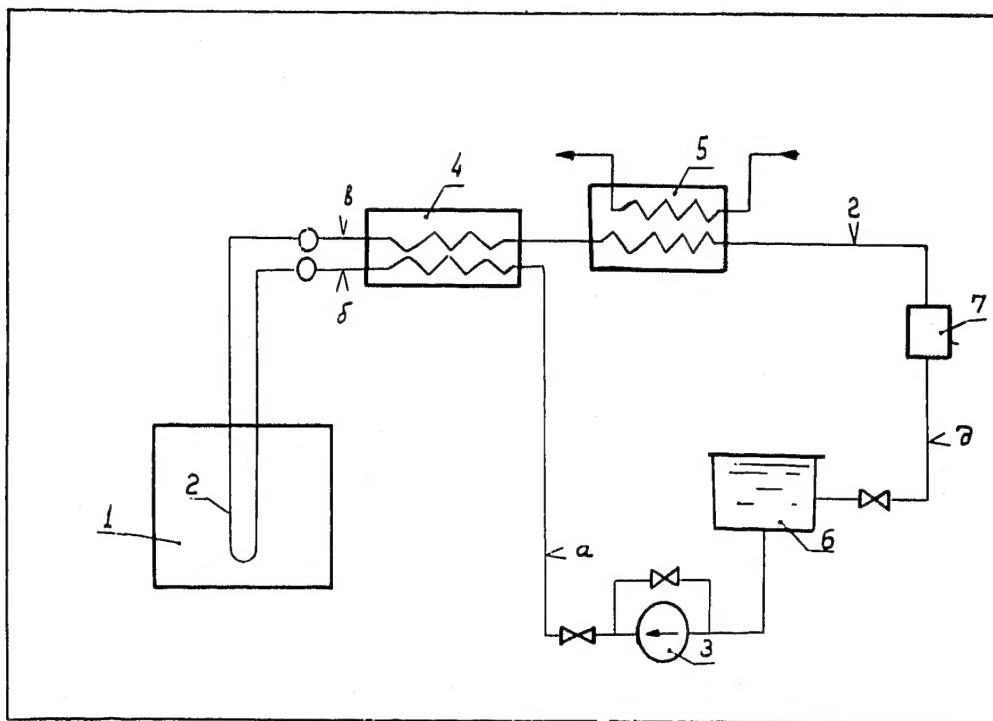4 – headers; 5 – sampling cooler.

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FIG.3. BASIC DIAGRAM OF A LOOP FOR WATER AND CHEMICAL STUDIES
1 — reactor; 2 — experimental ports; 3 — feed pump; 4 — regenerative heat exchanger; 5 — heat exchanger; 6 — pump tank; 7 — ion-exchange filters; 8 — sampling points (a,b,c,d,e).
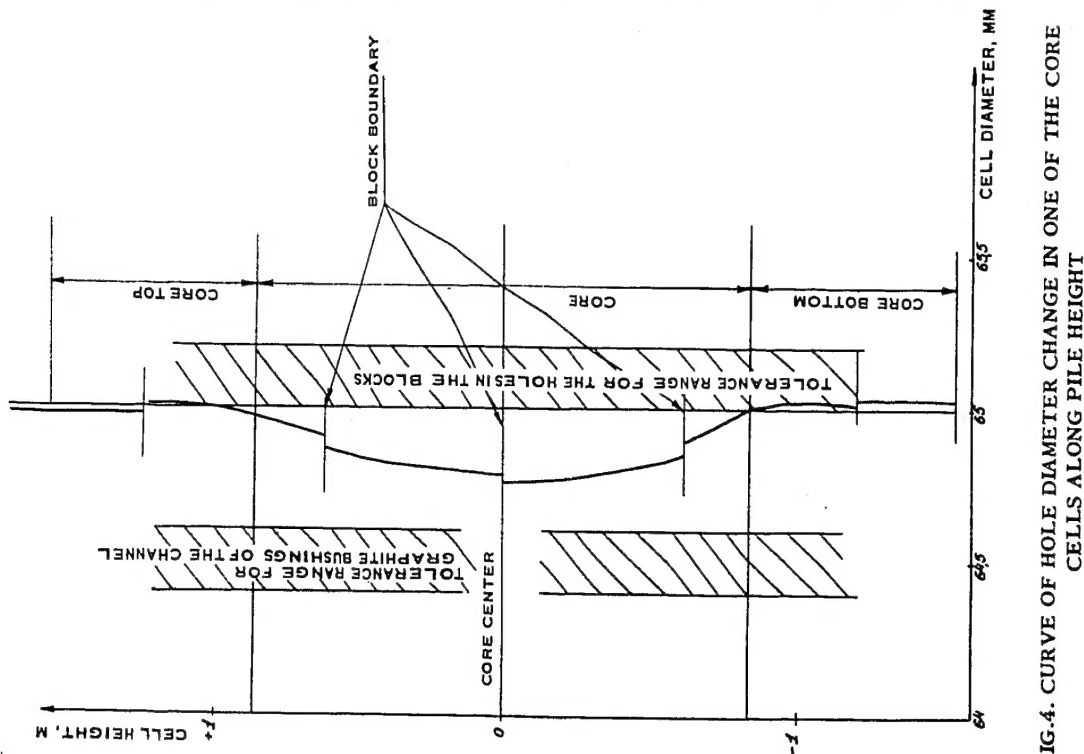

FIG.4. CURVE OF HOLE DIAMETER CHANGE IN ONE OF THE CORE CELLS ALONG PILE HEIGHT
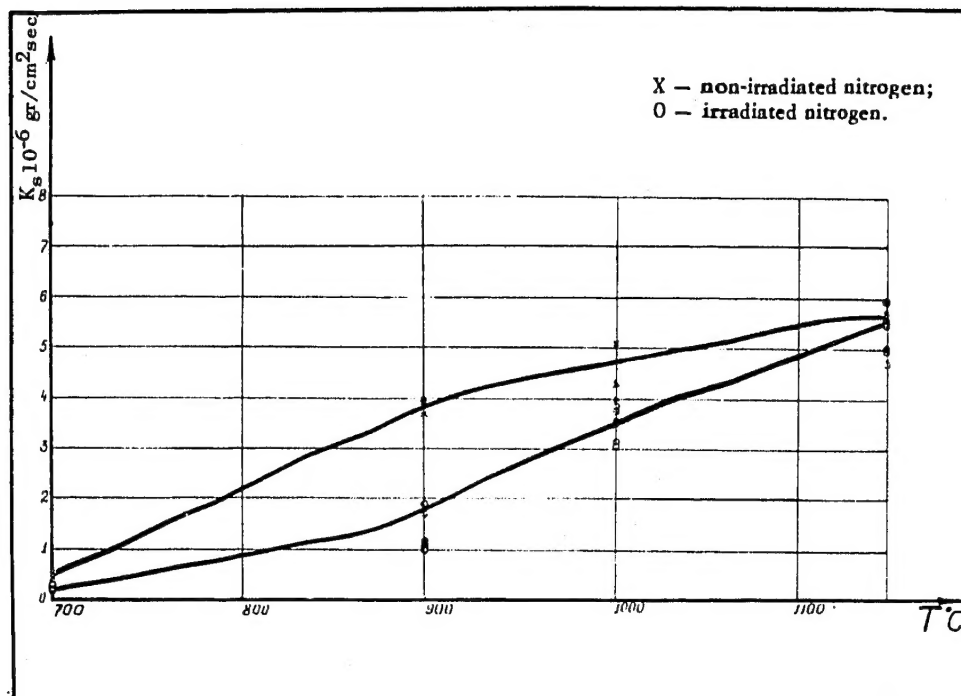
3 1 4

FIG.5. DEPENDENCY OF SPECIFIC VELOCITY QF EXIDATION ON TEMPERATURE

3 1 4

– 16 –